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Current status and technical description of Chinese $2 \times 250\,MW_{th}$ HTR-PM demonstration plant

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ABSTRACT

After the nuclear accidents of Three Mile Island and Chernobyl the world nuclear community made great efforts to increase research on nuclear reactors and to develop advanced nuclear power plants with much improved safety features. Following the successful construction and a most gratifying operation of the $10\,MW_{th}$ high-temperature gas-cooled test reactor (HTR-10), the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University has developed and designed an HTR demonstration plant, called the HTR-PM (high-temperature-reactor pebble-bed module). The design, having jointly been carried out with industry partners from China and in collaboration of experts worldwide, closely follows the design principles of the HTR-10.

Due to intensive engineering and R&D efforts since 2001, the basic design of the HTR-PM has been finished while all main technical features have been fixed. A Preliminary Safety Analysis Report (PSAR) has been compiled.

The HTR-PM plant will consist of two nuclear steam supply system (NSSS), so called modules, each one comprising of a single zone $250\,MW_{th}$ pebble-bed modular reactor and a steam generator. The two NSSS modules feed one steam turbine and generate an electric power of 210 MW.

A pilot fuel production line will be built to fabricate 300,000 pebble fuel elements per year. This line is closely based on the technology of the HTR-10 fuel production line.

The main goals of the project are two-fold. Firstly, the economic competitiveness of commercial HTR-PM plants shall be demonstrated. Secondly, it shall be shown that HTR-PM plants do not need accident management procedures and will not require any need for offsite emergency measures.

According to the current schedule of the project the completion date of the demonstration plant will be around 2013. The reactor site has been evaluated and approved; the procurement of long-lead components has already been started.

After the successful operation of the demonstration plant, commercial HTR-PM plants are expected to be built at the same site. These plants will comprise many NSSS modules and, correspondingly, a larger turbine.

1. Introduction

After the nuclear accidents in Three Mile Island and Chernobyl the world nuclear community started huge efforts to study and to develop advanced nuclear power systems with enhanced safety features. Advanced light water reactors (ALWR), which are lately labeled Generation-III reactors (and even GEN-III+ reactors), have been developed; e.g. the ABWR, the EPR, the System 80+, as well as the AP1000 and ESBWR. LWR-concepts which are beyond the ALWRs, such as IRIS by Westinghouse and PIUS by former ABB, were also developed. In the field of high-temperature gas-cooled reactors (HTGR), H. Reutler and G. Lohnert of German SIEMENS/INTERATOM proposed the modular concept of a $200\,MW_{th}$ HTR-MODUL in the early 1980s (Reutler and Lohnert, 1984). A very sophisticated, peculiar design of the HTR-MODUL guarantees that the maximum fuel temperature will never exceed the fuel's design limit for all accidents, such as e.g. "depressurized loss of coolant" or even "expulsion of all control rods", without needing any emergency cooling measures. According to statements made during an IAEA conference in 1992 these kind of reactors were called "nuclear power plants beyond the next generation". Since 2000 these concepts were labeled by the U.S. DOE as "Generation IV Nuclear Energy Systems" (NERAC, 2002).

HTGRs use helium as coolant and graphite as moderator as well as structural material. Its fuel elements contain thousands of very small "coated particles" which are embedded in a graphite matrix. At present, its core outlet helium temperature can reach 700–950 °C; even higher outlet temperatures are envisaged when

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the current research for better materials and improved fuel proves to be successful. Therefore, HTGR plants are even now capable of utilizing the high efficient and mature technologies of conventional fossil-fired power plants. For example, HTGR plants can achieve a thermal efficiency of 42% by even employing subcritical superheated steam turbines or reaching ~45% when supercritical steam turbines are installed. The efficiency could be improved even further when adopting direct helium gas turbines with recuperators or when choosing a combined cycle. In addition, the high-temperature heat sources provided by HTGRs can be used in many industrial processes to replace coal, oil or natural gas. Only some processes such as heavy oil recovery, hydrogen production, coal gasification and liquefaction will be mentioned here.

From the early 1960s, the United Kingdom, the United States and Germany began to research and develop HTGRs. In 1962, the U.K. and the European Community cooperated to build the first HTGR (Dragon) in the world, which provided a thermal power of 20 MW and achieved criticality in 1964; it used fuel in the form of graphite rod-bundles. Thereafter, Germany successively built two pebble-bed nuclear plants, the 45 MW_{th} test HTGR (AVR) and the 750 MW_{th} HTGR power plant (THTR-300), while the U.S constructed the 40 MW_e graphite fuel rod-bundle core (the Peach-Bottom test reactor) and the 330 MW_e Fort. St. Vrain power plant utilizing prismatic graphite fuel. Japan started the construction of a 30 MW_{th} high-temperature test reactor (HTTR) in 1991, which attained its first criticality in 1998 (IAEA-TECDOC-1198, 2001).

Intensive R&D on modular HTGRs has been performed in Germany and in the United States since the 1980s. SIEMENS/INTERATOM designed a 200 MW_{th} pebble-bed modular high-temperature gas-cooled reactor (HTR-MODUL), while General Atomics adopted the principles of the German HTR-MODUL and worked on a 350 MW_{th} prismatic fuel type modular high-temperature gas-cooled reactor (MHTGR), which later on was upgraded to a power output of 600 MW_{th} and connected to helium gas turbine, the GT-MHR. Since the middle of the 1990s, the company PBMR (pebble-bed modular reactor) of South Africa is developing a 400 MW_{th} pebble-bed modular reactor which also adopts the helium gas turbine cycle. The electrical power is envisaged to be 165 MW for an inlet-temperature of 500 °C and an outlet temperature of 900 °C. In the framework of Generation-IV reactors, the United States is planning to implement the Next Generation Nuclear Plant (NGNP) project around 2020 by constructing an HTGR demonstration plant for electricity and hydrogen production (Idaho National Laboratory, INL/EXT-07-12967, 2007).

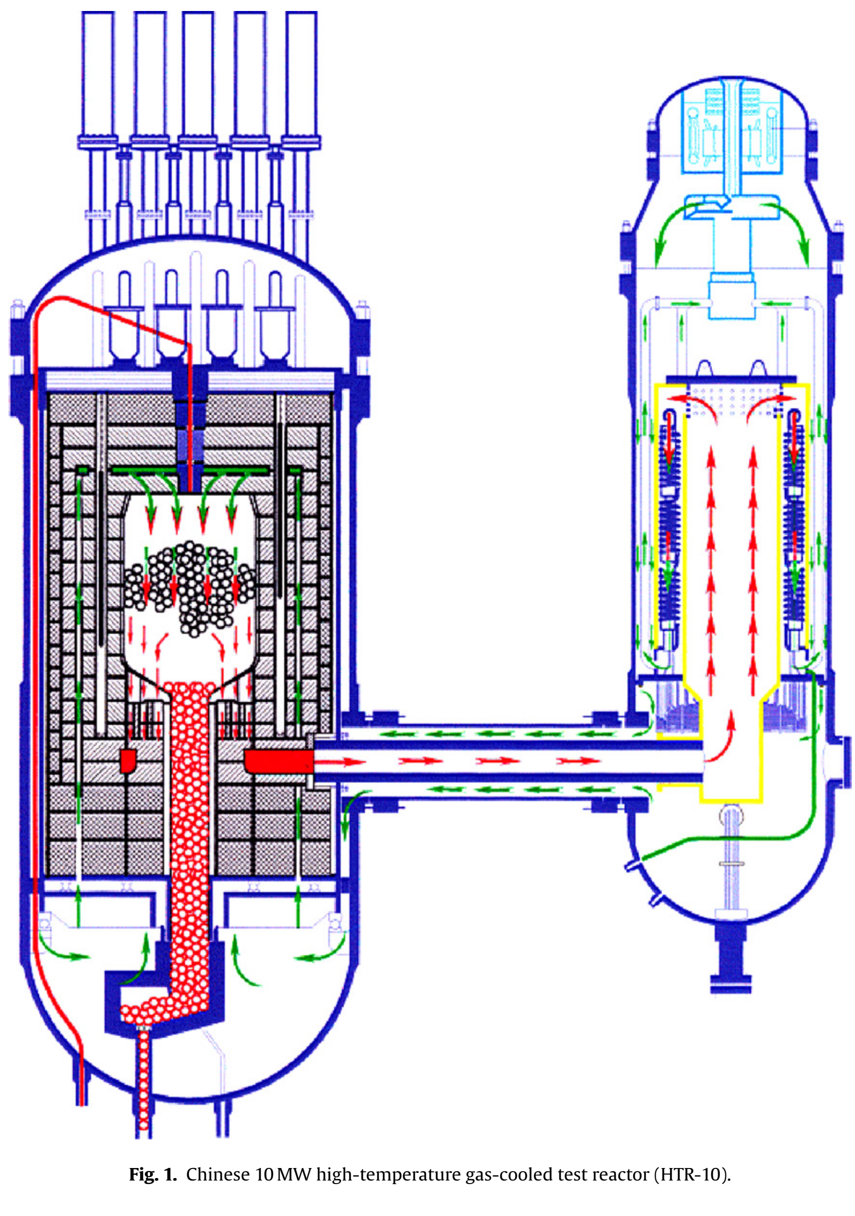

Fig. 1. Chinese 10 MW high-temperature gas-cooled test reactor (HTR-10).

In China, R&D of HTGRs began in the middle of the 1970s. From 1974 to 1985, the "Institute of Nuclear and New Energy Technology" (INET) of the Tsinghua University carried out some basic research on HTGR technology. After 1986, the R&D of HTGR technology was highly intensified by co-operating with the international HTGR community, especially with German institutions. During 1986–1990 eight key technical research topics, including 43 sub-tasks, were identified and systematical in-depth experimental studies were carried out. In 1992, China decided to construct the 10 MW_{th} high-temperature gas-cooled test reactor (HTR-10) as shown in Fig. 1 (Wu et al., 2002) at the INET-site in Beijing.

This project is considered to be the first tangible step of HTGR development in China. The main objectives of the HTR-10 were:

- to acquire the know-how to design, construct and operate HTGRs,
- to demonstrate the inherent safety features of the modular HTGR,
- to establish an irradiation and experimental facility for fuel elements and materials,
- to carry out R&D work for nuclear high-temperature process-heat applications.

In June 1995, the first concrete for the HTR-10 reactor building was poured. Finally, in December 2000, the HTR-10 reached its first criticality, while in January 2003 the HTR-10 had been successfully connected to the electric grid. For 72 h the reactor worked at full power. From April 2003 to September 2006 INET completed four experiments to confirm and verify claimed crucial inherent safety features of modular HTRs:

- loss of offsite power without any counter-measures,
- main helium blower shutdown without any counter-measures,
- loss of main heat sink without any counter-measures, and especially
- withdrawal of all rods without any counter-measures.

All these experiments were authorized, guided and supervised by the National Nuclear Safety Authority (NNSA).

During this R&D-period of the HTR-10, five significant achievements were obtained:

(1) *Manufacture of spherical coated particle fuel element*: the know-how of fabricating fuel elements for the HTR-10 was mastered. The free Uranium fraction could steadily be decreased to a value as low as 3×10^{-5}.
(2) *Corresponding technologies for pebble-bed HTGRs*: the technology of fuel element handling and spherical fuel element transportation by pulse pneumatic mechanism.
(3) *Helium process technologies*: such as helium sealing and purification, the lubrication for rotating equipments in a helium atmosphere, electrical insulation and rotor dynamics.
(4) *Domestic manufacture of key equipments for HTGRs*: this mainly covers the reactor pressure vessel, the steam generator pressure vessel, the hot gas duct, the once-through steam generator with helical tubes, the helium blower, the fuel handling equipments and reflector graphite components.
(5) *Successful development of fully digital reactor protection systems*.

The Chinese licensing authority, i.e. the National Nuclear Safety Authority, issued all required licensing documents by carefully and intensively reviewing the Preliminary Safety Analysis Report (PSAR), the Final Safety Analysis Report (FSAR) and other relevant supplementary documents. Thus, by licensing the HTR-10 the NNSA acquired large experience and knowledge of HTGRs.

A second step of HTGR-application in China had been started in 2001 (Zhang and Yu, 2002) when the high-temperature gas-cooled reactor pebble-bed module (HTR-PM) project was launched. The preliminary investment agreement was signed in December 2004 by "China Huaneng Group", by "China Nuclear Engineering and Construction Corporation" and by "Tsinghua Holding Corporation". In January 2006 the project named "Large Advanced Pressurized Water Reactor and High-Temperature Gas-cooled Reactor Nuclear Power Plants" became one of the 16 top priority projects of the "Chinese Science and Technology Plan" for the period 2006–2020. By February 2008 the implementation plan and the budget for the HTR-PM project was approved by the State Council of China. In November 2003, the "Chinergy Company" has been established and was designated to be the main contractor of the HTR-PM nuclear island, while in January 2007 the "Huaneng Shandong Shidao Bay Nuclear Power Company" was founded being the owner of the HTR-PM demonstration plant.

2. Significance and technical progress of the HTR-PM project

2.1. Major tasks of the HTR-PM technology in the Chinese market

The roles of the HTR-PM technology in the Chinese market are:

(1) *Alternative to LWRs in nuclear power*: the Chinese government has announced to have 40 GW_e nuclear power plants in operation and additional 18 GW_e in construction by 2020. The nuclear power capacity will be expanded to more than 100 GW_e between 2020 and 2040. The HTR-PM could be a supplement to larger LWR plants.
(2) *Alternative to oil and natural gas*: China is currently the second largest oil import country in the world. The HTR-PM could provide a high-temperature heat source for hydrogen production, for heavy oil thermal recovery, for coal gasification and liquefaction and for other industrial heat needs. The HTR would be a major nuclear solution for these purposes.
(3) *Next step of technology innovation after HTR-10*: the HTR-10 had been the first step of HTGR development and of advanced nuclear energy systems. The HTR-PM must – consequently – be the next step, otherwise the gained vast expertise and the large economic expenses during the last 20 years will be lost.

2.2. Main technical goals of the HTR-PM project

The HTR-PM should achieve the following technical goals:

(1) *Demonstration of inherent safety features*: the inherent safety features of modular HTGR power plants guarantees and requires that under all conceivable accident scenarios the maximum fuel element temperatures will never surpass its design limit temperature without employing any dedicated and special emergency systems (e.g. core cooling systems or special shutdown systems, etc.). This ensures that accidents (e.g. similar to LWRs core melting) are not possible so that not acceptable large releases of radioactive fission products into the environment will never occur.
(2) *Demonstration of economic competitiveness*: the first HTR-PM demonstration power plant will be supported by the Chinese government, so that the owner can always maintain the plant operation and obtain investment recovery. However, this government supported demonstration plant has to prove that a cost overrun during the construction period will be avoided and that the predicted smooth operation and performance will be kept. Hence, the demonstration plant must clearly demonstrate that follow-on HTR-PM plants will be competitive to LWR plants without any government support.

(3) *Confirmation of proven technologies*: in order to minimize the technical risks the successful experiences gained from the HTR-10 and from other international HTGR plants will be fully utilized in the HTR-PM project. The HTR-PM reactor design is very similar to the HTR-10. The turbine plant design will use the mature technology of super-heated steam turbines which is widely used in other thermal power plants. Besides, the manufacture of fuel elements will be based on the technology verified and proven during the HTR-10 project. In addition, the key systems and equipments of the plant will be rigorously tested in large-scale experimental rigs in order to guarantee the safety and reliability of all components. Furthermore, international mature technologies and successful experiences will be absorbed through international technical consultations.

(4) *Standardization and modularization*: the HTR-PM demonstration plant, consisting of two pebble-bed module reactors of combined $2 \times 250\,MW_{th}$ power, adopts the operation mode of two modules connected to only one steam turbine/generator set. This design allows to demonstrate the advantages and key benefits of employing and implementing a design of standardization and modularization. If the construction and operation of the HTR-PM demonstration plant proves to be successful, larger scale HTR-PM plants – using multiple-modules feeding one steam turbine only – will become a reality.

2.3. Technical progress of the HTR-PM

The technical research for the HTR-PM began in 2001. The main technical scheme of the nuclear island was finally fixed in 2006. The key technology research and engineering verifications are carried out according to elaborate plans. An HTGR engineering laboratory and a large helium engineering testing loop are under construction at INET. Here, the engineering verification experiments for the main component prototypes will be performed on large test rigs offsite the reactor.

The expected project construction period from pouring the first tank of concrete to generating electricity for the grid is scheduled to be 50 months. Although the workload of building, construction and installation is relatively clear and straight forward, the project schedule, nevertheless, leaves certain time margins allowing for possible uncertainties. The current plan aims for feeding electricity to the national power grid in 2013.

According to the requirements of the HTR-PM project, a fuel production line will be built soon having a capacity of producing 300,000 spherical fuel elements per year.

Finally, the HTR-PM project will establish the technical foundations to be able to realize Generation-IV nuclear energy system goals in the next stage, such as:

(1) *Largely enhanced safety features*: a successful HTR-PM will have already proven this technical target of Generation-IV nuclear energy systems.

(2) *Achieving outlet temperatures beyond 1000 °C [very high-temperature gas-cooled reactor (VHTR)]*: the reactor of current design and using current fuel element technologies has already the potential of realizing a gas outlet temperature of 950 °C. A further improvement of the fuel element performance is already foreseeable which will allow reaching this goal of attaining an outlet-temperature of 1000 °C.

(3) *Hydrogen production, use of helium turbine or supercritical steam turbine*: the current reactor design, verified by the HTR-PM, can readily be applied for the helium turbine or super-critical steam turbine or for the generation of large-scale production of hydrogen by nuclear energy.

3. Technical description of HTR-PM

3.1. Overall technical description

The HTR-PM deploys pebble-bed modular high-temperature gas-cooled reactors of 250 MW thermal power. Two reactor modules are coupled with two steam generators which are connected to one steam turbine-generator of 210 MW electric power. The reactor and the steam generator are installed inside two separate pressure vessels. The pressure vessels are assembled in a staggered, side-by-side arrangement and are connected by a horizontal coaxial hot gas duct. The primary pressure boundary consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV) and the hot gas duct pressure vessel (HDPV), which all are housed in a concrete shielding cavity as shown in Fig. 2.

The main helium blower is mounted on the upper part of the steam generator pressure vessel. The core inlet helium temperature was chosen to be 250 °C while the outlet helium temperature is 750 °C. The blower transfers the reactor heat to the steam generator, where high-pressure super-heated steam is produced which drives the steam turbine.

The ceramic structures surrounding the reactor core consist of the inner graphite reflector and outer carbon brick layers. The reactor core does not contain any fuel-free region or a graphite reflector in the center. The control rod channels are located in the side graphite reflector close to the core, while the returning cold helium is guided through borings in the outer part of the side reflector. The whole ceramic internals are installed inside a metallic core barrel, which itself is supported by the RPV. The metallic core barrel and the pressure vessel are protected against high temperatures from the core by the cold helium borings of the side reflector, which act like a shielding temperature screen.

The spherical fuel element with a diameter of 60 mm contains a multitude of ceramic coated particles. The coated fuel particles are uniformly embedded in a graphite matrix of 50 mm in diameter; while an outer fuel-free zone of pure graphite surrounds the spherical fuel zone for reasons of mechanical and chemical protection. Each spherical fuel element contains about 12,000 coated fuel particles. A coated fuel particle with a diameter of nearly 1.0 mm is composed of a UO_2 kernel of 0.5 mm diameter and three PyC layers and one SiC layer (TRISO). The heavy metal contained in each spherical fuel element is chosen to be 7.0 g. The design burn-up will be 90 GWd/tU, while the maximum fuel burn-up will not be in excess of 100 GWd/tU. In order to reach a fairly uniform distribution of fissile material throughout the whole core a “multi-pass” scheme of fuel circulation had been adopted.

In summarizing, the HTR-PM has the following important technical design features:

(1) By using spherical fuel elements containing TRISO coated particles one can assure that all relevant radioactive fission products will effectively be retained for at least 500 h when, during accidents, a maximum temperature of 1620 °C is not exceeded.

(2) The pebble-bed core design allows the spherical fuel elements to constantly pass though the core by gravity from up to down. This fuelling scheme avoids loading the core with excess reactivity. The elaborate reactor core design ensures that the fuel element temperature will never exceed the safety limit of 1620 °C for any operating or accident condition.

(3) The operation mode adopts continuous fuel loading and discharging: the fuel elements drop into the reactor core from the central fuel loading tube and are discharge through a fuel extraction pipe at the core bottom. Subsequently, the discharged fuel elements pass the burn-up measurement facility one by one. Depending on their state of burn-up they will either be discharged and transported into the spent fuel stor-

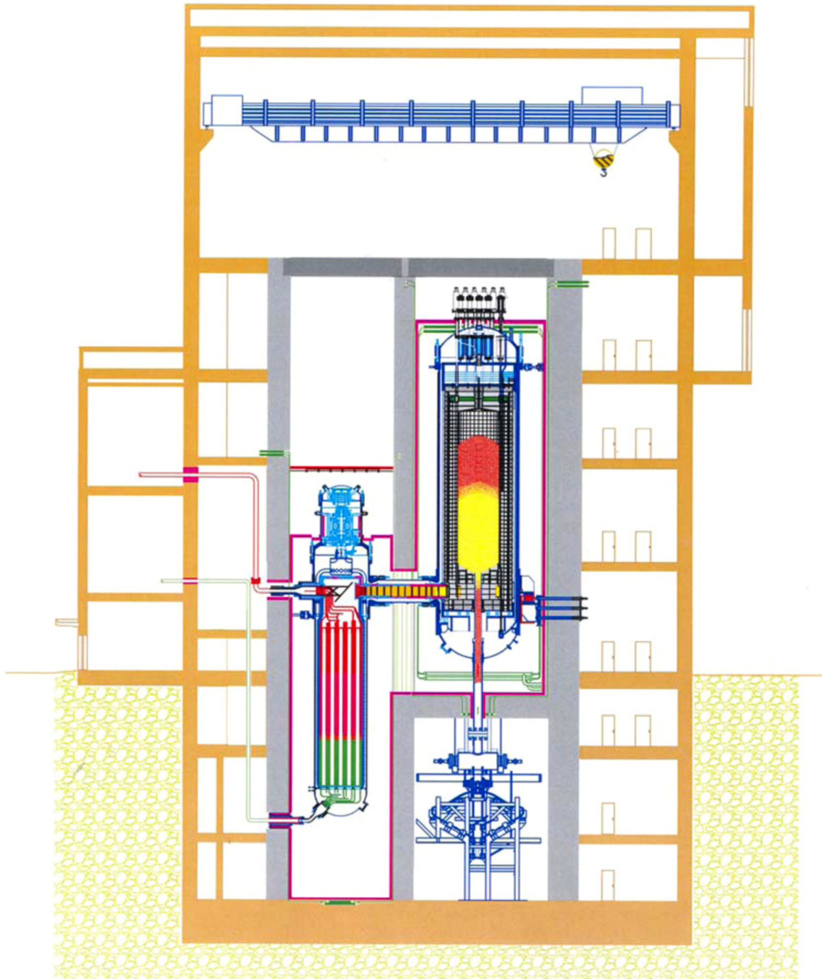

Fig. 2. Cross-section of the HTR reactor building.

age tank when having reached their design burn-up, or they will be re-inserted into the reactor to pass the core once again. The power distribution of the core depends on the number of passes one chooses. Obviously, the higher the number of passes is chosen the flatter will be the power distribution. This is favorable when regarding fuel element temperatures during accident conditions. On the other hand, a high number of fuel passes complicates the fuel handling devices as well as the complexity of the burn-up measurement facility.

(4) Two independent shutdown systems are installed: a control rod system and a small absorber sphere (SAS) system, both placed in holes of the graphitic side reflector. For shutdown purposes the rods and the small absorbers are released and drop into the reflector borings by gravity. This will improve the reliability of the shutdown systems.

(5) The active core zone is encased by a bulky layer of graphite and carbon bricks without metallic components. This ensures that the core internals can withstand and endure very high temperatures.

(6) The reactor core and the steam generator heat transfer bundles are installed in two different pressure vessels, which are connected by the hot gas duct pressure vessel. The primary pressure boundary comprises all three vessels. These vessels are all protected by the cold helium gas (250 °C). This ensures that moderate vessel temperatures are reached during reactor operation and in accident scenarios.

(7) All three primary loop pressure vessels (RPV, SGPV and the HDPV) are located in a concrete cavity, which protects the primary loop from external loads.

3.2. Main reactor plant equipments

The main systems and equipments of the reactor plant include the reactor internals, the control rod system and the small absorber sphere system, the reactor primary pressure vessels, the main helium blower and the steam generator.

The reactor internals consist of graphitic, carbonic and metallic components. The graphitic internals act primarily as the neutron reflector; in addition they provide the means to be able to arrange the helium flow channels and the absorber borings. The main function of the metallic internals is to support the graphite and carbon internals along with the ceramic structure of the pebble-bed core, and to pass various loads and forces to the reactor pressure vessel.

The control rod system and the small absorber sphere system are two independent control systems of reactivity. These two independent systems fulfill the requirements of diversity and redundancy. There are 8 control rods and 22 small absorber sphere units, both are located in the reflector region.

The three primary pressure vessels are composed of SA533-B steel as the plate material and (or) the 508-3 steel as the forging material. These materials meet the technical requirements of ASME-III-1-NB.

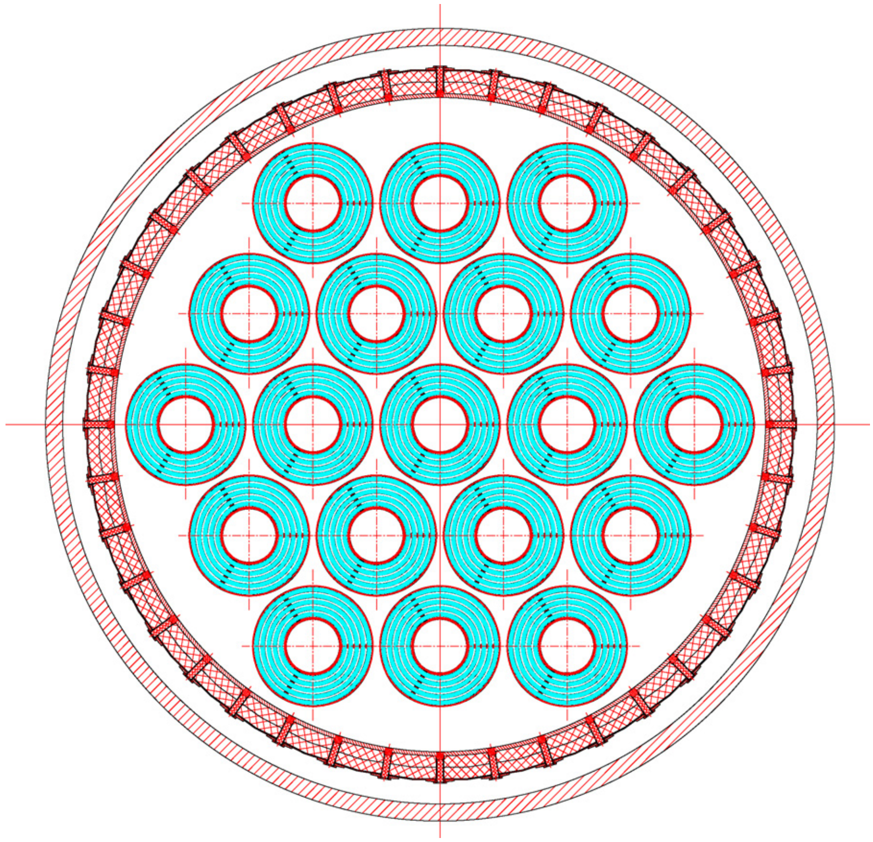

Fig. 3. Cross section of steam generator with 19 assemblies.

The main helium blower, designed as a vertical structure, is installed on the top of the steam generator inside the steam generator pressure vessel. The electric motor is mounted on an insertable assembly; the motor is driven by the converter outside of the pressure vessel. A magnetic bearing system is envisaged.

The steam generator consists of 19 separate helical tube assemblies; each assembly has 5 layers and includes 35 helical tubes, as shown in Figs. 2 and 3. To ensure two-phase flow stability, throttling apertures are installed at the entrance of all helical tubes. The assembly type design of the steam generator uses the experiences from the steam generator employed in the HTR-10. In-service inspection is possible. For full verification of the steam generator assembly full scale testing will be performed.

Due to the favorable temperature features of HTGRs, a superheated high-pressure steam turbine is adopted for the HTR-PM; these components exhibit high reliability and economical viability. Above all, there are mature experiences in the design, fabrication, serial manufacturing and operation of these systems and components in China.

3.3. Main technical parameters

The main technical parameters of the HTR-PM are presented in Table 1.

3.4. Key technical decisions

Before deciding on the design of the HTR-PM some fundamental decisions were made:

(1) *Steam turbine cycle or helium turbine cycle*: from 2001 to 2003, INET cooperated with the East China Electric Power Design Institute (ECEPDI) to carry out a pre-conceptual research for HTGR power plants. Three technical solutions were compared in detail; these included:
  (a) the conventional indirect steam turbine cycle,
  (b) the direct helium turbine arrangement, and
  (c) the indirect helium turbine arrangement.

Table 1
Main design parameters of the HTR-PM.

Parameter	Unit	Value
Rated electrical power	MW_e	210
Reactor total thermal power	MW_{th}	2×250
Designed life time	a	40
Average core power density	MW/m^3	3.22
Electrical efficiency	%	42
Primary helium pressure	MPa	7
Helium temperature at reactor inlet/outlet	°C	250/750
Fuel type		TRISO (UO_2)
Heavy metal loading per fuel element	g	7
Enrichment of fresh fuel element	%	8.9
Active core diameter	m	3
Equivalent active core height	m	11
Number of fuel elements in one reactor core		420,000
Average burn-up	GWd/tU	90
Type of steam generator		Once through helical coil
Main steam pressure	MPa	13.24
Main steam temperature	°C	566
Main feed-water temperature	°C	205
Main steam flow rate at the inlet of turbine	t/h	673
Type of steam turbine		Super high-pressure condensing bleeder turbine

It was found that the direct helium turbine technology had certain technical uncertainties. Thus, the conditions for constructing an industrial demonstration plant of this type were not ready yet. The following technical problems needed to be solved: the inspection technology of helium turbo-compressor blades under aggravating conditions of radioactivity deposition, the design and verification technology of RPV materials, the magnetic bearing technology, high-speed rotor dynamics and control technology, high efficiency recuperator-technology, to name only a few.

As for the R&D issue of helium turbine technology, INET is now in charge of the national research project named "10 MW High-Temperature Gas-cooled Reactor Helium Turbine System" (HTR-10GT). The goal of this project is to build a helium turbine electricity-generating system to study the key technical difficulties related to the HTGR helium turbine technology. The commercial-scale HTGR helium turbine cycle could only be realized in the future when a debugged, tested and verified HTR-PM reactor had been combined with mature and verified components of helium turbine technology.

In 2003 it was decided to use a steam turbine cycle for the HTR-PM project after the three above mentioned cycles had been intensively studied and scrutinized. The plant will have a design thermal efficiency of 42%.

(2) *Two-module reactors coupled with one steam turbine*: two different designs have been studied –
  (a) The first one was a 458 MW_{th} reactor with a two-zone annular core (Zhang et al., 2006). This kind of design adopts a fuel-free graphite reflector placed in the center of the pebble-bed core in order to increase the reactor thermal power as much as possible. However, our evaluations indicated that – for the time being – there are grave technical uncertainties in this design.
  (b) The second design consists of a 2×250 MW_{th} reactor plant; each reactor has a one-zone cylindrical core. The Chinese HTR-10 is a one-zone pebble-bed reactor in a side-by-side arrangement of rector and steam generator. Hence, the design of a 2×250 MW_{th} plant of two one-zone reactors is regarded as a mere up-scaling of the HTR-10 as prototype. Through the practices and experiences obtained by the design, construction and operation of the HTR-10, the technical uncertainties of this second design will be reduced decisively. Besides, the HTR-MODUL plant, designed by SIEMENS/INTERATOM and licensed by the German licensing authorities, also adopted the two-module reactors design and will be regarded as a reference.

(3) *Costs*: after having carried out the economic comparison of the above first two designs, it was clearly found that the specific cost differences are small (Zhang and Sun, 2007). The main reasons are as follows –
  (a) In order to reduce the helium flow resistance, the primary pressure of the 458 MW_{th} reactor had to be increased to 9.0 MPa. Also the diameter of the pressure vessel had to be enlarged. By contrast, the 250 MW_{th} reactor needs only a pressure of 7.0 MPa while having a smaller diameter of the pressure vessel. Therefore, the total weight of the primary pressure boundary components of the 2×250 MW_{th} reactors is only 14% higher than the weight of the primary pressure boundary of a 458 MW_{th} reactor.
  (b) Three trains of equipments for the fuel handling systems are required for the 458 MW_{th} reactor, while the 2×250 MW_{th} reactors need only two.
  (c) Since one has to take into account a necessary replacement of the central graphite reflector during the lifetime of the 458 MW_{th} plant, the reactor building for this design is higher and larger. The detailed comparison of the two designs is depicted in Table 2.

Table 2
Comparison of two HTR-PM designs.

	1 × 458 MW design	2 × 250 MW design
RPV weight	1	2 × 0.57
Graphite weight	1	2 × 0.60
Metallic reactor internals weight	1	2 × 0.86
Main blower power	1	2 × 0.57
Number of control rods	24	2 × 8
Number of small absorber sphere systems	8	2 × 22
Number of fuel handling system	3	2
Volume of reactor plant building	1	0.96
Number of reactor protection systems	1	2
Number of main control room	1	1
Helium purification systems	2 × 100%	2 × 100%
Fresh fuel and spent fuel systems	1 × 100%	1 × 100%
Emergency electrical systems	2 × 100%	2 × 100%

(4) *Other aspects*: the "module" concept has the following meaning –
  (a) aeveral identical modules are arranged to form a large plant, which is of benefit to standardization as well as reduction of manufacturing costs;
  (b) a relatively small nuclear power-output of a module reactor is indispensable when wanting to realize inherent safety features;
  (c) therefore, a modular nuclear power plant must consist of several or even many modules. However, the total plant capacity, but not the single module power, is pivotal and decisive for the economics of a power plant. Auxiliary systems, infrastructure and other indirect costs can be shared by all the modules.

The HTR-PM adopts a layout mode of two-module reactors coupled to one steam turbine in order to verify the feasibility and rationality of coupling a multitude of modules to one steam turbine. In addition, this kind of arrangement enables to test the sharing of the auxiliary systems. Furthermore, a 2×250 MW thermal power output coincides with the availability of 200 MW_e steam turbine products in the Chinese market.

4. Safety-performance and economics of the HTR-PM

4.1. Safety-performance of the HTR-PM

The HTR-PM will realize the following safety features:

(1) the radioactive inventory in the primary helium coolant is very small when the reactors are working at normal operation conditions. Even if this limited amount of radioactivity would be released into the environment following an accident, there is no need to take any emergency measures;
(2) for any reactivity accident or for a loss of coolant accident the rise of the fuel element temperature will not cause a significant additional release of radioactive substances from the fuel elements;
(3) the consequences of water or air ingress accidents depend on the quantity of such ingresses. The ingress processes and the associated chemical reactions are slow, and can readily be terminated within several dozens of hours (or even some days) by taking very simple actions.

The nuclear safety goal of the HTR-PM can be summarized follows:

The consequences of all conceivable accidents will not result in significant offsite radioactive impacts. The plant meets already the safety target of Generation-IV nuclear energy systems which stipulates: “eliminate the need for offsite emergency measures”. The same viewpoint is put down by IAEA in its report No. NS-R-1 “Safety of Nuclear Power Plants: Design”. Here it is expressively stated that “An essential objective is that the need for external intervention measures may be limited or even eliminated in technical terms, although such measures may still be required by national authorities.”

4.2. The economics of the HTR-PM

According to our investigations and regarding specific costs (Zhang and Sun, 2007), there is no significant difference between an HTR-PM plant and a PWR plant when the costs of infrastructure, R&D, project management, etc. are effectively shared in a commercial-scale, multiple-module HTR-PM plant. Compared with PWRs, inherently safe HTR-PM plants exhibit smaller power density, in total heavier PRVs and core internals, and higher specific cost. The other components of a nuclear power plant, however, depend upon the power to be generated, and no significant difference exists between PWRs and an HTR-PM plant. The reactor pressure vessel and the costs of reactor internals of a PWR accounts for only ∼2% of the total plant costs (including financial cost, from the practical data in Chinese PWR project, Zhang and Sun, 2007), so the cost increase from RPVs and reactor internals in HTR-PM has a limited impact. This limited impact will be compensated by simplification of the reactor auxiliary systems, the I&C and electrical systems, as well as by the benefit of mass production for the conventional island equipments, RPVs and reactor internals. In addition, it is expected that the costs of an HTR-PM plant will be further decreased through reducing the workload of design and engineering management, shortening construction schedules and lessening financial costs by making use of modularization.

In summing up it is expected that modular HTGR power plants will show to be economically competitive with PWRs due to the following reasons:

(1) simple systems;
(2) high operation temperature and the use of a high-pressure super-heated steam turbine-generator; this is similar to normal fossil power plants. Hence, a much higher thermal efficiency can be realized;
(3) multiple-module reactors coupled to one steam turbine-generator, sharing common auxiliary systems, and further reducing the costs through modularization and standardization for manufacture and construction;
(4) the operation mode of on-line continuous fueling will improve the availability of the power plant;
(5) the design burn-up of the fuel is expected to reach at least 100 GWd/t or even more; this will reduce the fuel cycle costs.

From our current knowledge and for Chinese market conditions we estimate the necessary budget excluding R&D and infrastructure costs for the first HTR-PM demonstration plant to be about 2000 USD/kW$_e$.

Of course, all these claims, drawn from our year-long analysis, must clearly be verified in detail. By successfully operating the HTR-PM in the very near future we are confident to reach these our claims.

5. Conclusions

On the basis of the HTR-10, the ongoing Chinese HTR-PM project is considered to be a decisive new step for the development of Chinese HTGR technology. Its main target is to finish building a pebble-bed HTR-PM demonstration plant of 210 MW$_e$ around 2013. Through the mutual efforts of all relevant scientific research organizations and industrial enterprises, and having the strong support of the Chinese government, the HTR-PM project will certainly play an important role in the world-wide development of Generation-IV nuclear energy technologies.

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